Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.54(f)

October 27, 2004
3F1004-05

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Crystal River Unit 3 – 60-Day Response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections"

Reference: NRC dated August 30, 2004, Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections"

Dear Sir:

Pursuant to 10 CFR 50.54(f), Florida Power Corporation, doing business as Progress Energy Florida, Inc., hereby submits the Crystal River Unit 3 (CR3) 60-Day response to NRC Generic Letter (GL) 2004-01.

The attachment to this letter provides the information requested in Option (a) of the GL. The Attachment concludes that inspection techniques applied during the previous Once-Through Steam Generator Tube Inspection, performed in October 2003, met the inspection requirements of the CR3 Improved Technical Specifications, 10 CFR 50, Appendix B requirements and the NRC position provided in the GL.

This letter establishes no new regulatory commitments.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/lvc

Attachment: Response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," Option (a)

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector

A115

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 27th day of October, 2004, by Dale E. Young

Janet Schroeder

Signature of Notary Public
State of Florida

JANET SCHROEDER
MY COMMISSION # DD 128063
EXPIRES: June 20, 2006
Bonded Thru Notary Public Underwriters

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT

Response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," Option (a)

NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," dated August 30, 2004, was sent to all holders of operating licenses for pressurized-water reactors (PWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel. The generic letter requested the following information within 60 days:

Requested Information

1. *Addressees should provide a description of the Steam Generator (SG) tube inspections performed at their plant during the last inspection. In addition, if they are not using SG tube inspection methods whose capabilities are consistent with the NRC's position, addressees should provide an assessment of how the tube inspections performed at their plant meet the inspection requirements of the Technical Specifications (TS) in conjunction with Criteria IX and XI of 10 CFR Part 50, Appendix B, and corrective action taken in accordance with Appendix B, Criterion XVI. This assessment should also address whether the tube inspection practices are capable of detecting flaws of any type that may potentially be present along the length of the tube required to be inspected and that may exceed the applicable tube repair criteria.*

Response

Background

Crystal River Unit 3 (CR3) has two Babcock & Wilcox (B&W) designed 177FA Once-Through Steam Generators (OTSGs). Each OTSG contains 15,531 sensitized Inconel-600 (I-600) tubes that have an outer diameter of 0.625 inch with a nominal wall thickness of 0.037 inch. Each tube is supported by 15 tube support plates (TSPs) that are 1.5-inches thick carbon steel and have trefoil broached holes, except for the 15th TSP, which has drilled holes for the 1,621 tubes at the outer periphery of the tube bundles. The upper and lower tube ends are roll-expanded to a minimum depth of 1.0 inch from the primary face of the tubesheet and a fillet weld exists between the primary face of the tubesheet and the tube end. A repair roll has been qualified for installation in the upper tubesheet or the lower tubesheet to repair indications of Primary Water Stress Corrosion Cracking (PWSCC) and/or Intergranular Attack (IGA). After installation, the repair roll becomes the new pressure boundary.

CR3 operates on approximately a 24-month fuel cycle. The CR3 steam generators had operated for 17.6 EFPY at the time of their last inspection in October 2003, which was the unit's 13th refueling outage (13R).

Previous Inspection Information

The CR3 13R OTSG tube eddy current inspection is summarized in Table 1. The steam generator tube inspection scope was governed by a number of sources, including the results of the CR3 degradation assessment and "Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) SG Examination Guidelines, Revision 6, Appendix H," which describes the performance demonstration requirements for eddy current techniques used for the examination of steam generator tubing and repairs. These criteria were used as the basis for determination of technique qualification. The in-service inspection of the OTSG tubes during 13R satisfied the requirements of the CR3 Improved Technical Specifications (ITS) Section 5.6.2.10.

In addition to the ITS inspection requirements, the CR3 degradation assessment evaluated the EPRI PWR SG Examination Guidelines in effect at the time of the inspection and available industry data for steam generators of similar design to determine potential damage mechanisms that may exist in the steam generators. Inspection techniques capable of detecting the potential and known degradation mechanisms, as identified in the CR3 13R Degradation Assessment, were employed in the respective areas.

For the lower tubesheet inspection, the kidney region was defined as the area on the lower tubesheet secondary face that had a sludge pile height of $\geq$1 inch as measured by eddy current inspection (ECT). The examination of the lower tubesheet (LTS) crevice area, outside of the kidney region and below LTS-8 inch in the kidney region, and the area outside the sludge pile region containing little sludge (<1 inch) was performed with a technique that did not meet all Appendix H requirements. Specifically, the probability of detection (POD) of 0.80 with a 90% confidence level using a data set of 11 or more flawed grading units. The sample set for the lower tube sheet region did not meet all the Appendix H requirements. However, CR3 did use a bobbin coil to examine the LTS crevice area with a reduced POD since the LTS is less susceptible to stress corrosion cracking due to the cooler temperature and service environment of an OTSG. The bobbin coil is acceptable to detect the expected degradation mechanism in the LTS crevice region, which is IGA. Since the bobbin coil technique has a reduced POD for detecting certain types of degradation, such as Stress Corrosion Cracking (SCC), in the crevice region, a sampling Motorized Rotating Coil (MRC) examination was performed in the kidney region and lower tube end roll region.

The ECT was performed by personnel qualified to the ASME Code Section XI, "Rules for In-service Inspection of Nuclear Power Plant Components," 1989 Edition, and to the requirements of EPRI PWR SG Examination Guidelines, Rev. 6, Appendix G, "Qualification of Nondestructive Examination Personnel for Analysis of Nondestructive Examination Data." The nondestructive examination procedures and equipment used to perform the ECT met the requirements of the ASME Code Sections XI and V, "Nondestructive Examination," 1989 Edition, as well as the requirements of the EPRI PWR SG Examination Guidelines, Revision 6. CR3 procedures were in place to verify and ensure that all personnel, equipment and inspection processes were qualified to the appropriate requirements and that the examination results were reviewed and documented to assure that the test requirements were satisfied.

Conclusion:

As previously discussed, CR3 performed an assessment to determine the types of degradation that potentially could occur along the length of a tube, ensure that appropriate inspection techniques were applied to detect potential degradation that may have been present, and to ensure that tube repairs were performed to maintain the integrity of the OTSGs. These measures ensured that the CR3 ITS requirements, 10 CFR Part 50, Appendix B Criteria IX, "Control of Special Processes," XI, "Test Control," and XVI, "Corrective Action," were satisfied. Detailed information regarding the 13R OTSG tube inservice inspection was provided to the NRC in Special Report 04-01: "Results of the Once-Through Steam Generator Tube Inservice Inspection Conducted During Refueling Outage 13" dated January 27, 2004.

Based on the information provided in Table 1 and the discussion above, the CR3 OTSG tube inspection approach/methods are in full compliance with the plant's ITS, 10CFR Part 50, Appendix B requirements, and the NRC's position as provided in Generic Letter 2004-01.

Table 1
CR3 13R OTSG Eddy Current Inspection
(October 2003)

Item	Steam Generator Region	Inspection Probe	Inspection Scope In-Service Tubes
1	Full Length of Tube (Note 1)	Bobbin	100% of all tubes in service; full length of all un-sleeved tubes
2	Dents ≥ 2.5 Volts (Note 2)	+Point^tm	34% of all dents LTS +4 inch to UTE 100% of the dents adjacent to Explosively Plugged Tubes
3	Sludge Pile / Lower Tube Sheet Crevice / Kidney Region (Note 3)	+Point^tm Pancake Coil	34% of the Sludge Pile, including dented tubes
4	Upper Tube Ends, Upper Original Roll Transition, or Lowest Repair Roll Transition (Note 4)	+Point^tm	100% of un-sleeved tubes
5	Lower Tube Ends, Lower Original Roll Transition, or Highest Repair Roll Transition (Note 4)	+Point^tm	100%
6	Lane and Wedge – 15^{th} TSP and Upper Tubesheet Face	+Point^tm Pancake	34% UTS and 15S of Un-sleeved tubes and One Tube Boundary Around Lane/Wedge Region Sleeved and Un-Sleeved Tubes
7	1^{st} Span B-0TSG tubes identified with IGA	+Point^tm High Frequency Bobbin	100% Recorded Indications and New Indications from Bobbin
8	Upper Tubesheet	+Point^tm	100% Recorded Indications and New Indications from Bobbin

Item	Steam Generator Region	Inspection Probe	Inspection Scope In-Service Tubes
9	Bobbin Indications (Note 5)	+Point^{tm} Pancake	100% Non-Quantifiable Indications (I-Codes), New Wear Indications, Impingement, and PLP
10	Alloy 690 Sleeves – Unexpanded Region	Bobbin	34% In-Service Sleeves
11	Alloy 690 Sleeves – Upper Roll and Lower Roll Expansion	+Point^{tm}	34% In-Service Sleeves
12	I-600 Roll Plug Expansion	Pancake	100% installed plugs
13	Possible Plugged Tube Sever	Inspect with Bobbin Coil and plug and stabilize downstream adjacent tubes	100% of Tubes at Risk

Notes for Table 1:

Note 1 Full-length of the tube is defined as: from point of entry to point of exit. The previously existing tube and tube roll, outboard of a new roll area in the tube sheet, is excluded from future periodic inspection requirements because it is no longer part of the pressure boundary after a repair roll is installed. For tubes with sleeves, the portion of the tube without the sleeve was also included.

Note 2 CR3 does not use the "ding" nomenclature; all indications of mechanical tube deformation are called "dents."

Note 3 The region inspected was: 4 inch above to 8 inch below of the Lower Tubesheet (Cold Leg) Secondary Face.

Note 4 Inspection of tube ends and original roll transition is not required in a tube that has a repair roll installed. +Point^{tm} inspection to 1 inch beyond the inboard roll transition is required.

Note 5 +Point^{tm} probe inspection was performed on bobbin coil indications of possible degradation, all recorded permeability variation indications, all recorded pilgering indications, and all recorded dent indications. New wear indications on bobbin coil inspection were confirmed with +Point^{tm}.

Acronym List

ID	Inside Diameter
IGA	Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
LTE	Lower Tube End
LTSF	Lower Tubesheet Face
MBM	Manufacturing Burnish Mark
OD	Outside Diameter
ODIGA	Outside-Diameter Intergranular Attack
ODSCC	Outside-Diameter Stress Corrosion Cracking
PLP	Possible Loose Part
PWSCC	Primary Water Stress Corrosion Cracking
S	Support Plate
SCC	Stress Corrosion Cracking
TSP	Tube Support Plate
UTE	Upper Tube End
UTS	Upper Tubesheet

Requested Information

2. *If addressees conclude that full compliance with the TS in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B, requires corrective actions, they should discuss their proposed corrective actions (e.g., changing inspection practices consistent with the NRC's position or submitting a TS amendment request with the associated safety basis for limiting the inspections) to achieve full compliance. If addressees choose to change their TS, the staff has included in the attachment suggested changes to the TS definitions for a tube inspection and for plugging limits to show what may be acceptable to the staff in cases where the tubes are expanded for the full depth of the tubesheet and where the extent of the inspection in the tubesheet region is limited.*

Response:

As stated in response to Requested Information 1, for CR3 all areas of potential and non-active damage mechanisms, as determined by the CR3 13R degradation assessment, were inspected using inspection techniques capable of detecting known and potential degradation mechanisms. Therefore, CR3 is in full compliance with ITS in conjunction with 10 CFR Part 50, Appendix B Criteria.

Requested Information

3. *For plants where SG tube inspections have not been or are not being performed consistent with the NRC's position on the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, the licensee should submit a safety assessment (i.e., a justification for continued operation based on maintaining tube structural and leakage integrity) that addresses any differences between the licensee's inspection practices and those called for by the NRC's position. Safety assessments should be submitted for all areas of the tube required to be inspected by the TS where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used, and should include the basis for not employing such inspection techniques. The assessment should include an evaluation of (1) whether the inspection practices rely on an acceptance standard (e.g., cracks located at least a minimum distance of x below the top of the tube sheet, even if these cracks cause complete severance of the tube) which is different from the TS acceptance standards (i.e., the tube plugging limits or repair criteria), and (2) whether the safety assessment constitutes a change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the joint. If the safety assessment constitutes a change to the method of evaluation under 10 CFR 50.59, the licensee should determine whether a license amendment is necessary pursuant to that regulation.*

Response

Not applicable based on response to Requested Information 2 above.